Southern Nuclear
Operating Company, Inc.

SOUTHERN COMPANY
Energy to Serve Your World

February 16, 2010

Docket Nos.: 50-321
50-366

NL-10-0246

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
ISI Program Alternative HNP-ISI-ALT-09

Ladies and Gentlemen:

Pursuant to 10 CFR 50.55a(a)(3)(ii), Southern Nuclear Operating Company (SNC) hereby requests approval of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 2001 Edition through 2003 Addenda for Hatch Nuclear Plant (HNP). SNC requests NRC approval of Alternative HNP-ISI-ALT-09, Version 1, which proposes VT-2 visual examinations during the system leakage test at a pressure lower than the Code required pressure following repair and replacement activities performed subsequent to the leakage test required by Table IWB-2500-1, Category B-P Item B15.10. SNC requests approval of the alternative for the remainder of the fourth ISI interval, which expires on December 31, 2015.

This alternative is requested due to the fact that compliance with the specified requirements would result in hardship without a compensating increase in the level of quality and safety. According to 10 CFR50.55a(a)(3), alternatives to the requirements of paragraph 50.55a(g) may be used, when authorized by the NRC, if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The ASME Code, Section XI, Table IWA-5210-1, "Reference Paragraphs for System Pressure Tests and Visual Examination" requires the performance of a system pressure test in accordance IWA-4500 for all Class 1 pressure boundary components following repair/replacement activities. IWA-4500 requires a system leakage test in accordance with IWA-5000, as supplemented by 10 CFR 50.55a(b)(2)(xxvi), in particular paragraph IWA-5211(a). IWA-5211(a) references Table IWB-2500-1, Category B-P which references IWB-5220 for test parameters. IWB-5221(a) requires that the system leakage test be conducted at a pressure not less than the nominal operating pressure associated with 100% rated reactor power.

HNP-ISI-ALT-09 proposes a system leakage test at a pressure lower than the Code required pressure for: (1) repair and replacement activities performed subsequent to plant startup following the Code required Class 1 System Leakage Test at the end of each refueling outage, and (2) for leakage testing following repair and replacement activities not associated with a refueling outage.

An outage is scheduled to begin on April 5, 2010 for HNP Unit 2 to replace two leaking safety relief valve (SRV) pilots and a SRV main body. One of the two replacement pilots is associated with the SRV main body replacement. It was recently determined that one SRV was experiencing internal leakage within the main body. Because of this unforeseen condition, SNC has conservatively decided to replace that SRV main body and its associated topworks during the outage scheduled to begin on April 5, 2010. At the present time, it is believed that there is no Class 1 pressure boundary leakage associated with the main body that is scheduled to be replaced. These repair/replacement activities are required to be pressure tested in accordance with IWB-5221(a), which requires a pressure of 1045 psig without the approval of this proposed alternative. Approval of this alternative is needed to perform visual leakage examination at a lower pressure. Performing the leakage test at a lower pressure reduces the temperatures and radiation levels in the drywell and would provide for a higher quality visual examination. The proposed alternative for pressure testing at 920 psig would detect leakage if the bolted connection was not leak tight.

The NRC has approved a similar relief request for Pilgrim Nuclear Power Station. This approval was contained in the NRC letter of June 29, 2006 (TAC NO. MC8286), which approved the Pilgrim Relief Request PRR No. 2.

SNC respectfully requests expedited approval of this proposed alternative on or before April 9, 2010, to facilitate the Hatch Unit 2 SRV outage of 2010.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

Mark J. Ajluni

M. J. Ajluni
Manager - Nuclear Licensing

MJA/PAH/lac

Enclosure: Request for Alternative HNP-ISI-ALT-09, Version 1

cc: <u>Southern Nuclear Operating Company</u>
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Mr. D. R. Madison, Vice President – Hatch
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<u>U. S. Nuclear Regulatory Commission</u>
Mr. L. A. Reyes, Regional Administrator
Ms. D. N. Wright, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

Enclosure

Edwin I. Hatch Nuclear Plant
ISI Program Alternative HNP-ISI-ALT-09

SOUTHERN NUCLEAR OPERATING COMPANY
HNP-ISI-ALT-09, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

<table>
<tr><td>Plant Site-Unit:</td><td>Edwin I. Hatch Nuclear Plant - Units 1 and 2</td></tr>
<tr><td>Interval-Interval Dates:</td><td>4th ISI Interval, January 1, 2006 through December 31, 2015</td></tr>
<tr><td>Requested Date for Approval and Basis:</td><td>Approval is requested by April 9, 2010 to support testing during the planned Unit 2 shutdown that begins on April 5, 2010.</td></tr>
<tr><td>ASME Code Components Affected:</td><td>Class 1 pressure retaining components which require a VT-2 examination for leakage after the Table IWB-2500-1, Category B-P, Item B15.10 system leakage has been performed or subsequent to a repair/replacement activity performed during an unplanned shutdown.</td></tr>
<tr><td>Applicable Code Edition and Addenda:</td><td>ASME Section XI Code, 2001 Edition through the 2003 Addenda</td></tr>
<tr><td>Applicable Code Requirements:</td><td>1. IWA-4540(a) requires a hydrostatic or system leakage test, in accordance with IWA-5000, for repair/replacement activities performed by welding or brazing on a pressure retaining boundary prior to, or as part of, returning to service.
2. IWB-5221(a) requires the system leakage test to be conducted at a pressure not less than the nominal pressure associated with 100% rated reactor power.</td></tr>
<tr><td>Reason for Request:</td><td>Relief is requested from the test pressure requirement of IWB-5221(a) (i.e., 1045 psig) on the basis of impracticality as cited below.
• Component repair or replacement may be required due to leakage identified during performance of the Class 1 system leakage test. Such repair/replacement activities may require VT-2 leakage examination which would be performed during unit startup.
• It is sometimes necessary to rework and re-examine mechanical joint connections after completion of the Class 1 system leakage test. Examples include:
  o Safety Relief Valves required to be replaced due to observed leakage during the Class 1 System Leakage Test.
  o Leaking mechanical joint connections discovered during the system leakage test that require repair/replacement prior to unit startup.
  o Control Rod Drive mechanisms that require repair or change-out after the Class 1 System Leakage Test.
• VT-2 leakage examination inside the drywell (primary containment) is not feasible at the nominal operating pressure of 1045 psig during start-up because of adverse radiation levels and high ambient and component temperatures.
• Nominal operation pressure (i.e., 1045 psig) is not achieved until approximately 24-hours after initial startup due to:
  o Control Rod Drive withdrawal limitations and the associated gradual increases in pressure and temperature.</td></tr>
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Technical Specification required Pressure versus Temperature limitations.
Main steam line piping, turbine control and stop valve warming requirements.
Main turbine warming requirements.
Steam driven safety system (i.e., HPCI and RCIC) surveillance requirements required prior to declaring operable.

VT-2 leakage examination at nominal operating pressure utilizing the normal start-up sequence would subject personnel to adverse conditions due to high temperature levels resulting from several hours of incremental pressure increases and resultant increases in drywell temperature.

For unit startup:

Control Rods are withdrawn to achieve initial core criticality.
Control Rod withdrawal continues, core power increases, and reactor coolant system (RCS) pressure and temperature increase.
Technical Specification Nil-Ductility Temperature limits for the RCS pressure versus temperature must be followed.
Core power control stabilizes at approximately 15% power and steam generation is adequate to support HPCI and RCIC surveillance testing and operation of the main feedwater pump(s).
Reactor power increase is halted at this level to allow for surveillance tests and piping and component warming which requires approximately 24-hours.

RCA nominal operating pressure results in drywell ambient temperatures that require special safety precautions such as ice vests and cool air supply lines for personnel performing the VT-2 examinations.
These adverse conditions could also compromise the quality of the leakage examination due to the hardship imposed on examination personnel.
Inspection inside the drywell is not feasible above approximately 5% reactor power because of the associated radiation levels, reactor coolant system temperature and ambient air temperatures.
920 psig is the nominal pressure associated with steam driven safety system operability (e.g., High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC)) and the pressure at which final system surveillances are performed.
The Technical Specifications require inerting the drywell with nitrogen within 24-hours after reaching 15% thermal power which prevents personnel entry.

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<td>Proposed Alternative and Basis for Use:</td>
<td>Plant Hatch will perform the required VT-2 leakage examination for any repair/ replacement activities performed after completion of the Table IWB-2500-1, Category B-P, Item B15.10 Class 1 System Leakage Test or subsequent to a repair/ replacement activity performed during an unplanned shutdown at a reactor steam dome pressure of ≥ 920 psig. Entry during startup, at this low power level (i.e., ≤5% and ≥920 psig), is achievable since component and ambient temperatures have not reached unsafe levels.</td>
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	Disposition of any observed leakage will consider the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100% rated reactor power (i.e., 1045 psig) and the actual reactor pressure when the examination was performed. Since the reactor coolant system (RCS) boundary is subjected to a leakage test and visual examination at nominal operating pressure (i.e., 1045 psig) near the end of every refueling outage, a leakage test and visual examination performed at 920 psig for repair/replacement of components ≤ 1" NPS and mechanical joint connections provides adequate assurance of structural and pressure boundary integrity. Therefore, this proposed alternative should be granted pursuant to 10 CFR 50.55a(a)(3)(ii).
Duration of Proposed Alternative:	The 4th ISI Interval, beginning June 1, 2009 and ending December 31, 2015.
References:	Entergy Nuclear Northeast Pilgrim Nuclear Power Station 4th 10-Year Interval ISI Program Relief Request PRR-2 NRC TAC NO. MC8286 dated June 29, 2006
Status:	Awaiting NRC approval.